June 14, 1988

Docket No. 50-389

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Mr. W. F. Conway
Vice President-Nuclear
Florida Power and Light Company
Post Office Box 14000
Juno Beach, Florida 33408

Dear Mr. Conway:

SUBJECT: ST. LUCIE UNIT 2 - ISSUANCE OF AMENDMENT RE: PRESSURE/TEMPERATURE LIMITS AND LOW TEMPERATURE OVERPRESSURE PROTECTION (TAC NO. 66709)

The Commission has issued the enclosed Amendment No. 31 to Facility Operating License No. NPF-16 for the St. Lucie Plant, Unit No. 2. This amendment consists of changes to the Technical Specifications (TS) in partial response to your application dated November 27, 1987, as supplemented by letters dated May 4 and 20, 1988.

This amendment changes the Reactor Coolant System Pressure/Temperature (P/T) limit figures to be effective up to six (6.0) effective full power years of operation. The amendment also changes the Technical Specifications dealing with overpressure protection systems because they are linked with the new P/T limit figures. The applicable bases sections are also changed to reflect the above changes.

A copy of the related Safety Evaluation (SE) is also enclosed. It provides the basis for issuing the changes specified above, and it also states that our review is continuing on two issues. The first issue deals with a set of P/T limit figures valid beyond 6.0 effective full power years based upon the method used by Florida Power and Light (FPL) to predict the reference nil ductility transition temperature shift as a function of fast neutron irridation. The second issue deals with action times to be used when no required overpressure protection devices are operable (LCO 3.4.9.3).

With regard to the first issue, the staff has agreed to conduct a detailed review, separate from this amendment, of a Combustion Engineering Report submitted by FPL with the November 27, 1987 application, to determine whether it justifies the FPL method for predicting delta RT_{NDT}. We will contact FPL if additional information is needed to complete our review. As discussed in the SE, the staff's current evaluation of the changes proposed in this amendment is based, in part, on the methods recommended in Regulatory Guide 1.99, Revision 2 (Draft).

With regard to the second issue, the staff has agreed to conduct a separate review of the adequacy of the action times associated with LCO 3.4.9.3. FPL believes that these times, which are in keeping with those in the corresponding section of the Combustion Engineering Standard Technical Specifications, NUREG-0212, Revision 2, are not realistic. Accordingly, we will be contacting FPL for additional information and then schedule a formal review.

The Notice of Issuance of this amendment will be included in the Commission's biweekly Federal Register notice.

Sincerely,

Original signed by

E. G. Tourigny, Project Manager
Project Directorate II-2
Division of Reactor Projects-I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 31 to NPF-16
2. Safety Evaluation

cc w/enclosures:
See next page

*SEE PREVIOUS PAGE FOR CONCURRENCE

LA:PDII-2*	PE:PDII-2*	PM:PDII-2*	D:PDII-2*	SRXB*
DMiller	JSchiffgens:bd	ETourigny	HBerkow	WHodges
05/04/88	05/09/88	05/09/88	05/09/88	05/11/88
EMTB*	OGC			
CCheng				
05/09/88	06/ /88			

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

FLORIDA POWER & LIGHT COMPANY

ORLANDO UTILITIES COMMISSION OF

THE CITY OF ORLANDO, FLORIDA

AND

FLORIDA MUNICIPAL POWER AGENCY

DOCKET NO. 50-389

ST. LUCIE PLANT UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 31
License No. NPF-16

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Florida Power & Light Company, et al. (the licensee), dated November 27, 1987, as supplemented May 4 and 20, 1988, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, Facility Operating License No. NPF-16 is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and by amending paragraph 2.C.2 to read as follows:

   2. <u>Technical Specifications</u>

      The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 31 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Herbert N. Berkow

Herbert N. Berkow, Director
Project Directorate II-2
Division of Reactor Projects-I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 14, 1988

ATTACHMENT TO LICENSE AMENDMENT NO. 31

TO FACILITY OPERATING LICENSE NO. NPF-16

DOCKET NO. 50-389

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change. The corresponding overleaf pages are also provided to maintain document completeness.

Remove Pages	Insert Pages
1-4	1-4
3/4 4-3	3/4 4-3
3/4 4-5	3/4 4-5
3/4 4-10	3/4 4-10
3/4 4-29	3/4 4-29
3/4 4-30	3/4 4-30
3/4 4-31	3/4 4-31a
---	3/4 4-31b
3/4 4-32	3/4 4-32
3/4 4-33	3/4 4-33
3/4 4-35	3/4 4-35
3/4 4-36	3/4 4-36
---	3/4 4-37a
3/4 4-38*	3/4 4-38*
B3/4 4-1	B3/4 4-1
B3/4 4-3	B3/4 4-3
B3/4 4-8	B3/4 4-8
B3/4 4-9	B3/4 4-9
B3/4 4-10	B3/4 4-10
B3/4 4-11	B3/4 4-11

*There is no change to this page. It is included to maintain document completeness.

DOSE EQUIVALENT I-131

1.10 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, " Calculation of Distance Factors for Power and Test Reactor Sites."

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

1.11 $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

ENGINEERED SAFETY FEATURES RESPONSE TIME

1.12 The ENGINEERED SAFETY FEATURES RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays where applicable.

FREQUENCY NOTATION

1.13 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.1.

GASEOUS RADWASTE TREATMENT SYSTEM

1.14 A GASEOUS RADWASTE TREATMENT SYSTEM is any system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system offgases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

IDENTIFIED LEAKAGE

1.15 IDENTIFIED LEAKAGE shall be:

a. Leakage (except CONTROLLED LEAKAGE) into closed systems, such as pump seal or valve packing leaks that are captured, and conducted to a sump or collecting tank, or

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE, or

c. Reactor Coolant System leakage through a steam generator to the secondary system.

1.16 LOW TEMPERATURE RCS OVERPRESSURE PROTECTION RANGE

The LOW TEMPERATURE RCS OVERPRESSURE PROTECTION RANGE is that operating condition when (1) the cold leg temperature is less than or equal to that specified in Table 3.4-3 for the applicable operating period, and (2) the Reactor Coolant System is not vented to containment by an opening of at least 3.58 square inches.

MEMBER(S) OF THE PUBLIC

1.17 MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the licensee, its contractors or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational or other purposes not associated with the plant.

OFFSITE DOSE CALCULATION MANUAL (ODCM)

1.18 The OFFSITE DOSE CALCULATION MANUAL shall contain the current methodology and parameters used in the calculation of offsite doses due to radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm/trip setpoints, and shall include the Radiological Environmental Monitoring Sample point locations.

OPERABLE - OPERABILITY

1.19 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s), and when all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

OPERATIONAL MODE - MODE

1.20 An OPERATIONAL MODE (i.e. MODE) shall correspond to any one inclusive combination of core reactivity condition, power level and average reactor coolant temperature specified in Table 1.2.

PHYSICS TESTS

1.21 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and (1) described in Chapter 14.0 of the FSAR, (2) authorized under the provisions of 10 CFR 50.59, or (3) otherwise approved by the Commission.

HOT SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.1.3 At least two of the loop(s)/train(s) listed below shall be OPERABLE and at least one Reactor Coolant and/or shutdown cooling loops shall be in operation.*

a. Reactor Coolant Loop 2A and its associated steam generator and at least one associated Reactor Coolant pump,**

b. Reactor Coolant Loop 2B and its associated steam generator and at least one associated Reactor Coolant pump,**

c. Shutdown Cooling Train 2A,

d. Shutdown Cooling Train 2B.

APPLICABILITY: MODE 4.

ACTION:

a. With less than the above required Reactor Coolant and/or shutdown cooling loops OPERABLE, immediately initiate corective action to return the required loops to OPERABLE status as soon as possible; if the remaining OPERABLE loop is a shutdown cooling loop, be in COLD SHUTDOWN within 30 hours.

b. With no Reactor Coolant or shutdown cooling loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required coolant loop to operation.

---

*All Reactor Coolant pumps and shutdown cooling pumps may be de-energized for up to 1 hour provided (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

**A Reactor Coolant pump shall not be started with two idle loops and one or more of the Reactor Coolant System cold leg temperatures less than or equal to that specified in Table 3.4-3 for the applicable operating period unless the secondary water temperature of each steam generator is less than 40°F above each of the Reactor Coolant System cold leg temperatures.

SURVEILLANCE REQUIREMENTS

4.4.1.3.1 The required Reactor Coolant pump(s), if not in operation, shall be determined to be OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.3.2 The required steam generator(s) shall be determined OPERABLE by verifying the secondary side water level to be $\geq$ 10% indicated narrow range level at least once per 12 hours.

4.4.1.3.3 At least one Reactor Coolant or shutdown cooling loop shall be verified to be in operation and circulating Reactor Coolant at least once per 12 hours.

REACTOR COOLANT SYSTEM

COLD SHUTDOWN - LOOPS FILLED

LIMITING CONDITION FOR OPERATION

3.4.1.4.1 At least one shutdown cooling loop shall be OPERABLE and in operation*, and either:

a. One additional shutdown cooling loop shall be OPERABLE#, or

b. The secondary side water level of at least two steam generators shall be greater than 10% indicated narrow range level.

APPLICABILITY: MODE 5 with Reactor Coolant loops filled##.

ACTION:

a. With one of the shutdown cooling loops inoperable and with less than the required steam generator level, immediately initiate corrective action to return the inoperable shutdown cooling loop to OPERABLE status or to restore the required steam generator level as soon as possible.

b. With no shutdown cooling loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required shutdown cooling loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.4.1.1 The secondary side water level of at least two steam generators when required shall be determined to be within limits at least once per 12 hours.

4.4.1.4.1.2 At least one shutdown cooling loop shall be determined to be in operation and circulating reactor coolant at least once per 12 hours.

*The shutdown cooling pump may be de-energized for up to 1 hour provided 1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and 2) core outlet temperature is maintained at least 10°F below saturation temperature.

#One shutdown cooling loop may be inoperable for up to 2 hours for surveillance testing provided the other shutdown cooling loop is OPERABLE and in operation.

##A Reactor Coolant pump shall not be started with two idle loops and one or more of the Reactor Coolant System cold leg temperatures less than or equal to that specified in Table 3.4-3 for the applicable operating period unless the secondary water temperature of each steam generator is less than 40°F above each of the Reactor Coolant System cold leg temperatures.

COLD SHUTDOWN - LOOPS NOT FILLED

LIMITING CONDITION FOR OPERATION

3.4.1.4.2 Two shutdown cooling loops shall be OPERABLE# and at least one shutdown cooling loop shall be in operation.*

APPLICABILITY: MODE 5 with reactor coolant loops not filled.

ACTION:

a. With less than the above required loops OPERABLE, within 1 hour initiate corrective action to return the required loops to OPERABLE status as soon as possible.

b. With no shutdown cooling loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and within 1 hour initiate corrective action to return the required shutdown cooling loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.4.2 At least one shutdown cooling loop shall be determined to be in operation and circulating reactor coolant at least once per 12 hours.

#One shutdown cooling loop may be inoperable for up to 2 hours for surveillance testing provided the other shutdown cooling loop is OPERABLE and in operation.

*The shutdown cooling pump may be deenergized for up to 1 hour provided (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

3/4.4.3 PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.3 The pressurizer shall be OPERABLE with a minimum water level of greater than or equal to 27% indicated level and a maximum water level of less than or equal to 68% indicated level and at least two groups of pressurizer heaters capable of being powered from 1E buses each having a nominal capacity of at least 150 kW.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one group of the above required pressurizer heaters inoperable, restore at least two groups to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With the pressurizer otherwise inoperable, be in at least HOT STANDBY with the reactor trip breakers open within 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.3.1 The pressurizer water volume shall be determined to be within its limits at least once per 12 hours.

4.4.3.2 The capacity of each of the above required groups of pressurizer heaters shall be verified to be at least 150 kW at least once per 92 days.

4.4.3.3 The emergency power supply for the pressurizer heaters shall be demonstrated OPERABLE at least once per 18 months by verifying that on an Engineered Safety Features Actuation test signal concurrent with a loss of offsite power:

a. the pressurizer heaters are automatically shed from the emergency power sources, and

b. the pressurizer heaters can be reconnected to their respective buses manually from the control room after resetting of the ESFAS test signal.

3/4.4.4 PORV BLOCK VALVES

LIMITING CONDITION FOR OPERATION

3.4.4 Each Power Operated Relief Valve (PORV) Block valve shall be OPERABLE. No more than one block valve shall be open at any one time.

APPLICABILITY: MODES 1, 2, and 3.#

ACTION:

a. With one or more block valve(s) inoperable, within 1 hour either restore the block valve(s) to OPERABLE status or close the block valve(s) and remove power from the block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With both block valves open, close one block valve within 1 hour, otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. The provisions of specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.4 Each block valve shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel unless the block valve is closed with power removed in order to meet the requirements of Action a. or b. above.

#When the RCS cold leg temperature is above the LOW TEMPERATURE RCS OVERPRESSURE PROTECTION RANGE of Table 3.4-3.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.9.1 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figures 3.4-2, 3.4-3 and 3.4-4 during heatup, cooldown, criticality, and inservice leak and hydrostatic testing.

APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operations or be in at least HOT STANDBY within the next 6 hours and reduce the RCS T_{avg} and pressure to less than 200°F and 500 psia, respectively, within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.1.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.9.1.2 The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, at the intervals required by 10 CFR 50 Appendix H in accordance with the schedule in Table 4.4-5. The results of these examinations shall be used to update Figures 3.4-2, 3.4-3 and 3.4-4.

FIGURE 3.4-2

ST. LUCIE-2 P/T LIMITS, 6 EFPY
HEATUP AND CORE CRITICAL

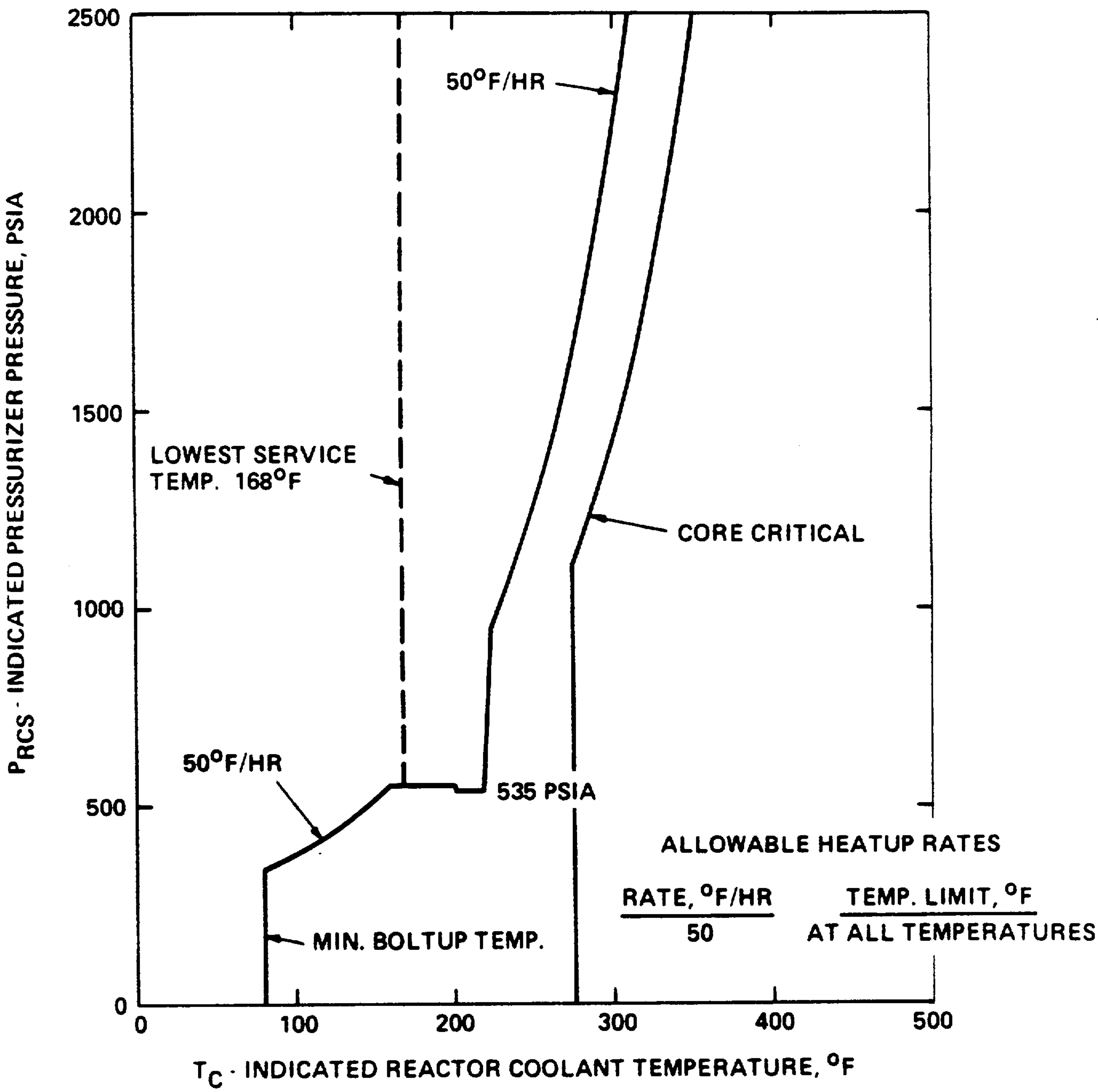


FIGURE 3.4-3

ST. LUCIE-2 P/T LIMITS, 6 EFPY
COOLDOWN AND INSERVICE TEST

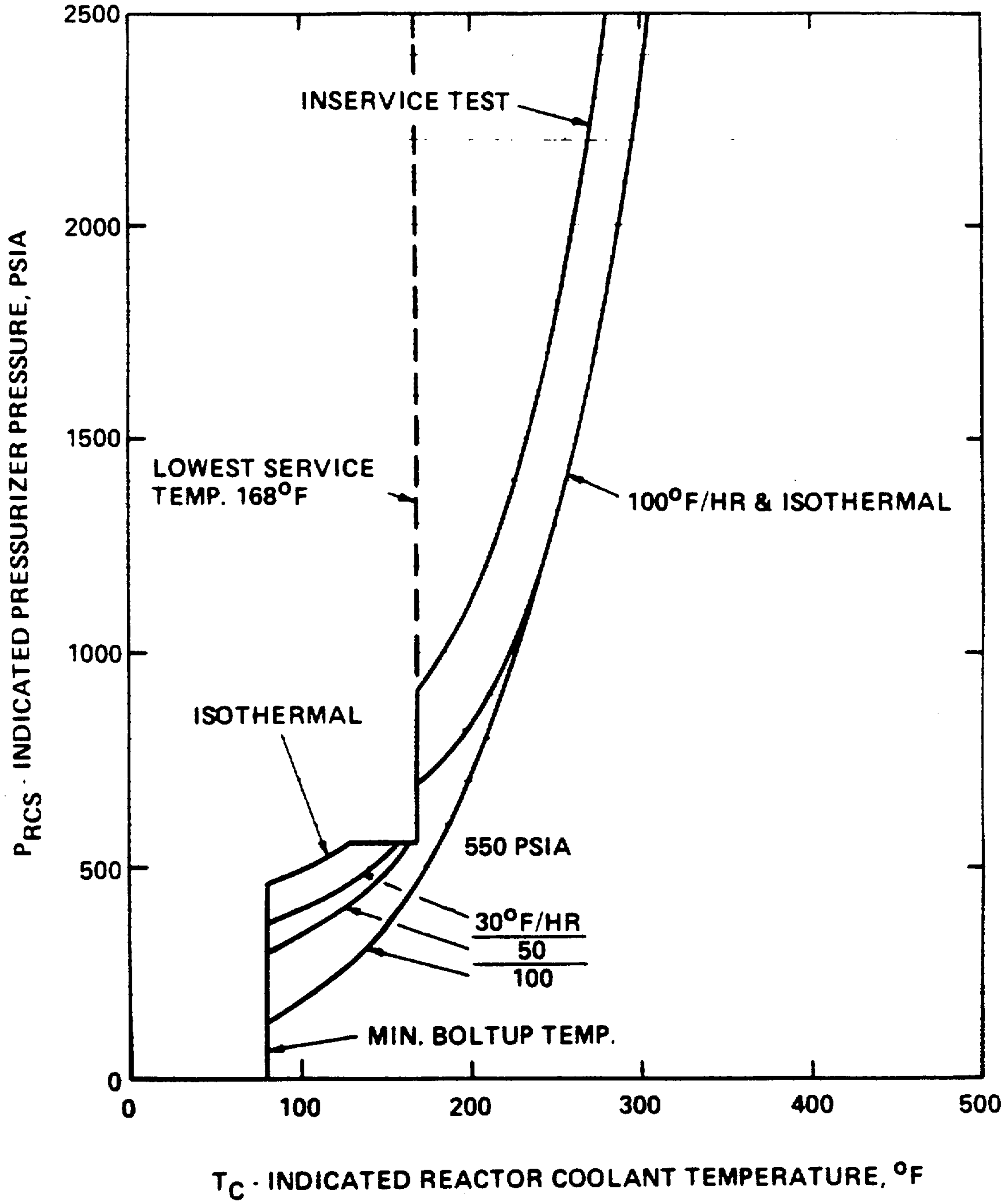
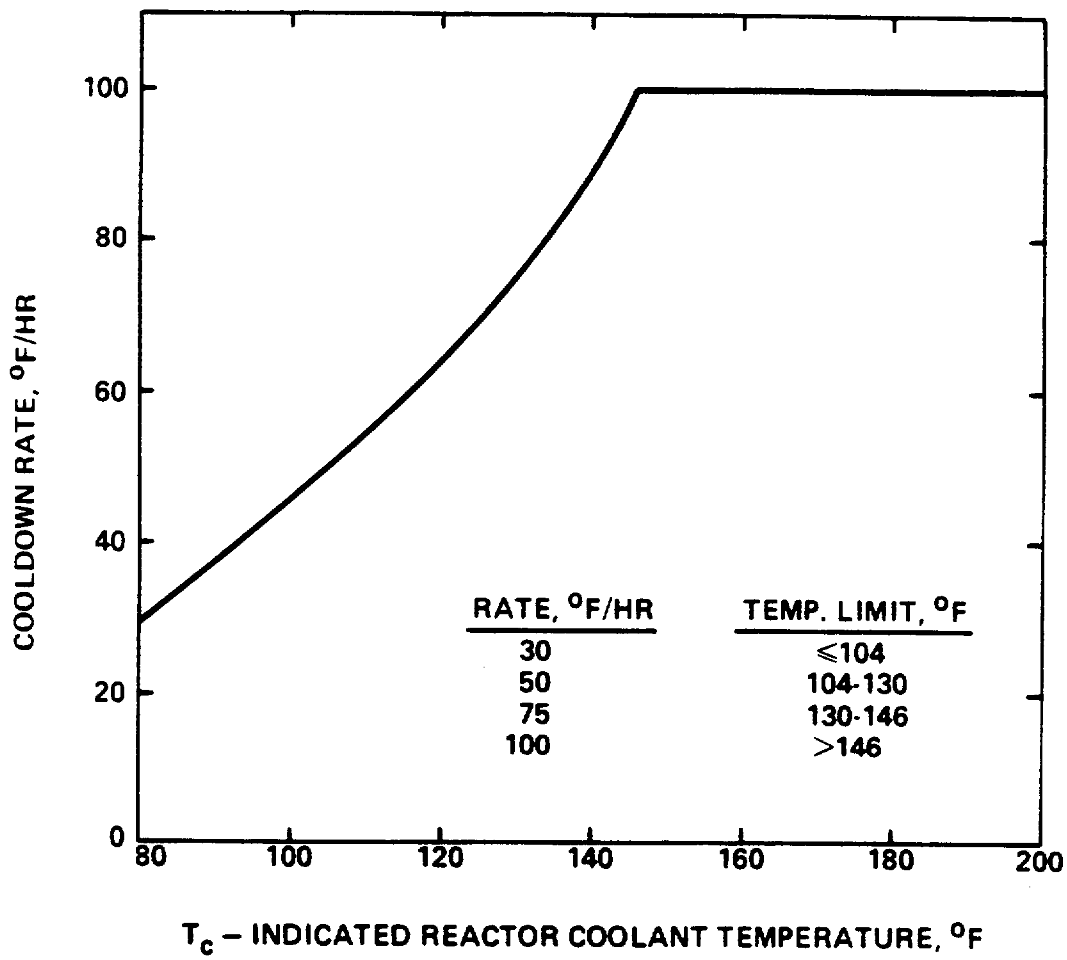


FIGURE 3.4-4
ST. LUCIE-2 P/T LIMITS, 6 EFPY
MAXIMUM ALLOWABLE COOLDOWN RATES



RATE, °F/HR	TEMP. LIMIT, °F
30	≤104
50	104-130
75	130-146
100	>146

NOTE: A MAXIMUM COOLDOWN RATE OF 100°F/HR IS ALLOWED AT ANY TEMPERATURE ABOVE 146°F

TABLE 4.4-5

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM - WITHDRAWAL SCHEDULE

CAPSULE NUMER	VESSEL LOCATION	LEAD FACTOR	WITHDRAWAL TIME (EFPY)
1	83°	$\leq$1.5	1.0
2	97°	$\leq$1.5	24.0
3	104°	$\leq$1.5	STANDBY
4	263°	$\leq$1.5	12.0
5	277°	$\leq$1.5	STANDBY
6	284°	$\leq$1.5	STANDBY

PRESSURIZER HEATUP/COOLDOWN LIMITS

LIMITING CONDITION FOR OPERATION

3.4.9.2 The pressurizer temperature shall be limited to:

a. A maximum heatup of 100°F in any 1-hour period, and

b. A maximum cooldown of 200°F in any 1-hour period.

APPLICABILITY: At all times.

ACTION:

With the pressurizer temperature limits in excess of any of the above limits, restore the temperature to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the pressurizer; determine that the pressurizer remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the pressurizer pressure to less than 500 psig within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.2 The pressurizer temperatures shall be determined to be within the limits at least once per 30 minutes during system heatup or cooldown.

OVERPRESSURE PROTECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.9.3 Unless the RCS is depressurized and vented by at least 3.58 square inches, at least one of the following overpressure protection systems shall be OPERABLE:

a. Two power-operated relief valves (PORVs) with a lift setting of less than or equal to 470 psia and with their associated block valves open. These valves may only be used to satisfy low temperature overpressure protection (LTOP) when the RCS cold leg temperature is greater than the temperature listed in Table 3.4-4.

b. Two shutdown cooling relief valves (SDCRVs) with a lift setting of less than or equal to 350 psia.

c. One PORV with a lift setting of less than or equal to 470 psia and with its associated block valve open in conjunction with the use of one SDCRV with a lift setting of less than or equal to 350 psia. This combination may only be used to satisfy LTOP when the RCS cold leg temperature is greater than the temperature listed in Table 3.4-4.

APPLICABILITY: MODES 3#, 4#, 5 and 6.

ACTION:

a. With either a PORV or an SDCRV being used for LTOP inoperable, restore at least two overpressure protection devices to OPERABLE status within 7 days or:

   1. Depressurize and vent the RCS with a minimum vent area of 3.58 square inches within the next 8 hours; OR

   2. Be at a temperature above the LOW TEMPERATURE RCS OVERPRESSURE PROTECTION RANGE of Table 3.4-3 within the next 8 hours.

b. With none of the overpressure protection devices being used for LTOP OPERABLE, within the next eight hours either:

   1. Restore at least one overpressure protection device to OPERABLE status or vent the RCS; OR

   2. Be at a temperature above the LOW TEMPERATURE RCS OVERPRESSURE PROTECTION RANGE of Table 3.4-3.

#With cold leg temperature within the LOW TEMPERATURE RCS OVERPRESSURE PROTECTION RANGE of Table 3.4-3.

LIMITING CONDITION FOR OPERATION

ACTION (Continued):

c. In the event either the PORVs, SDCRVs or the RCS vent(s) are used to mitigate a RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs, SDCRVs or vent(s) on the transient and any corrective action necessary to prevent recurrence.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.9.3.1 Each PORV shall be demonstrated OPERABLE by:

a. In addition to the requirements of Specification 4.0.5, operating the PORV through one complete cycle of full travel at least once per 18 months.

SURVEILLANCE REQUIREMENTS (Continued)

b. Performance of a CHANNEL FUNCTIONAL TEST on the PORV actuation channel, but excluding valve operation, within 31 days prior to entering a condition in which the PORV is required OPERABLE and at least once per 31 days thereafter when the PORV is required OPERABLE.

c. Performance of a CHANNEL CALIBRATION on the PORV actuation channel, at least once per 18 months.

d. Verifying the PORV isolation valve is open at least once per 72 hours when the PORV is being used for overpressure protection.

4.4.9.3.2 The RCS vent(s) shall be verified to be open at least once per 12 hours* when the vent(s) is being used for overpressure protection.

*Except when the vent pathway is provided with a valve which is locked, sealed, or otherwise secured in the open position, then verify these valves open at least once per 31 days.

TABLE 3.4-3

LOW TEMPERATURE RCS OVERPRESSURE PROTECTION RANGE

Operating Period, EFPY	Cold Leg Temperature, F° During Heatup	During Cooldown
4 $\geq$ 6	$\leq$ 313	$\leq$ 304

TABLE 3.4-4

MINIMUM COLD LEG TEMPERATURE FOR PORV USE FOR LTOP

Operating Period EFPY	T_{cold}, F° During Heatup	T_{cold}, F° During Cooldown
4 $\geq$ 6	156	179

REACTOR COOLANT SYSTEM

3/4.4.10 REACTOR COOLANT SYSTEM VENTS

LIMITING CONDITION FOR OPERATION

3.4.10 At least one Reactor Coolant System vent path consisting of two vent valves and one block valve powered from emergency buses shall be OPERABLE and closed at each of the following locations:

a. Pressurizer steam space, and

b. Reactor vessel head.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With one of the above Reactor Coolant System vent paths inoperable, STARTUP and/or POWER OPERATION may continue provided the inoperable vent path is maintained closed with power removed from the valve actuator of all the vent valves and block valves in the inoperable vent path; restore the inoperable vent path to OPERABLE status within 30 days, or, be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With both Reactor Coolant System vent paths inoperable, maintain the inoperable vent paths closed with power removed from the valve actuators of all the vent valves and block valves in the inoperable vent paths, and restore at least one of the vent paths to OPERABLE status within 72 hours or be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.10.1 Each Reactor Coolant System vent path shall be demonstrated OPERABLE at least once per 18 months by:

1. Verifying all manual isolation valves in each vent path are locked in the open position.

2. Cycling each vent valve through at least one complete cycle of full travel from the control room.

3. Verifying flow through the Reactor Coolant System vent paths during venting.

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with both reactor coolant loops and associated reactor coolant pumps in operation, and maintain DNBR above 1.20 during all normal operations and anticipated transients. In MODES 1 and 2 with one reactor coolant loop not in operation, this specification requires that the plant be in at least HOT STANDBY within 1 hour.

In MODE 3, a single reactor coolant loop provides sufficient heat removal capability for removing decay heat; however, single failure considerations require that two loops be OPERABLE.

In MODE 4, and in MODE 5 with reactor coolant loops filled, a single reactor coolant loop or shutdown cooling loop provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops (either shutdown cooling or RCS) be OPERABLE.

In MODE 5 with reactor coolant loops not filled, a single shutdown cooling loop provides sufficient heat removal capability for removing decay heat; but single failure considerations, and the unavailability of the steam generators as a heat removing component, require that at least two shutdown cooling loops be OPERABLE.

The operation of one reactor coolant pump or one shutdown cooling pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reductions will, therefore, be within the capability of operator recognition and control.

The restriction on starting a reactor coolant pump in MODES 4 and 5, with two idle loops and one or more RCS cold leg temperatures less than or equal to that specified in Table 3.4-3 for the applicable operating period is provided to prevent RCS pressure transients, caused by energy additions from the secondary system from exceeding the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients by (1) sizing each PORV to mitigate the pressure transient of an inadvertent safety injection actuation in a water-solid RCS with pressurizer heaters energized, (2) restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 40°F above each of the RCS cold leg temperatures, (3) using SDCRVs to mitigate RCP start transients and the transients caused by inadvertent SIAS actuation and charging water, and (4) rendering one HPSI pump inoperable when the RCS is at low temperatures.

3/4.4.2 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2750 psia. Each safety valve is designed to relieve 212,182 lbs per hour of saturated steam at the valve setpoint. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating shutdown cooling loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization. In addition, the Overpressure Protection System provides a diverse means of protection against RCS overpressurization at low temperatures.

SAFETY VALVES (Continued)

During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its safety limit of 2750 psia. The combined relief capacity of these valves is sufficient to limit the system pressure to within its Safety Limit of 2750 psia following a complete loss of turbine generator load while operating at RATED THERMAL POWER and assuming no reactor trip until the first Reactor Protective System trip setpoint (Pressurizer Pressure-High) is reached (i.e., no credit is taken for a direct reactor trip on the loss of turbine) and also assuming no operation of the pressurizer power-operated relief valve or steam dump valves.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Vessel Code.

3/4.4.3 PRESSURIZER

An OPERABLE pressurizer provides pressure control for the Reactor Coolant System during operations with both forced reactor coolant flow and with natural circulation flow. The minimum water level in the pressurizer assures the pressurizer heaters, which are required to achieve and maintain pressure control, remain covered with water to prevent failure, which could occur if the heaters were energized uncovered. The maximum water level in the pressurizer ensures that this parameter is maintained within the envelope of operation assumed in the safety analysis. The maximum water level also ensures that the RCS is not a hydraulically solid system and that a steam bubble will be provided to accommodate pressure surges during operation. The steam bubble also protects the pressurizer code safety valves against water relief. The requirement to verify that on an Engineered Safety Features Actuation test signal concurrent with a loss of offsite power the pressurizer heaters are automatically shed from the emergency power sources is to ensure that the non-Class 1E heaters do not reduce the reliability of or overload the emergency power source. The requirement that a minimum number of pressurizer heaters be OPERABLE enhances the capability to control Reactor Coolant System pressure and establish and maintain natural circulation.

3/4.4.4 PORV BLOCK VALVES

The power-operated relief valves (PORVs) and steam bubble function to relieve RCS pressure during all design transients up to and including the design step load decrease with steam dump. Operation of the PORVs in conjunction with a reactor trip on a Pressurizer Pressure-High signal minimizes the undesirable opening of the spring-loaded pressurizer code safety valves. The opening of the PORVs fulfills no safety-related function and no credit is taken for their operation in the safety analysis for MODE 1, 2, or 3.

Each PORV has a remotely operated block valve to provide a positive shutoff capability should a relief valve become inoperable. Since it is impractical and undesirable to actually open the PORVs to demonstate their reclosing, it becomes necessary to verify OPERABILITY of the PORV block valves to ensure the capability to isolate a malfunctioning PORV. As the PORVs are pilot operated and require some system pressure to operate, it is impractical to test them with the block valve closed.

The PORVs are sized to provide low temperature overpressure protection (LTOP). Since both PORVs must be OPERABLE when used for LTOP, both block valves will be open during operation within the LTOP range. As the PORV capacity required to perform the LTOP function is excessive for operation in MODE 1, 2, or 3, it is necessary that the operation of more than one PORV be precluded during these MODES. Thus, one block valve must be shut during MODES 1, 2, and 3.

The applicability of this technical specification to only a part of MODE 3 is due to the LTOP range slightly overlapping MODE 3 in the operating period beyond 15 EFPY. Both block valves will be open during operation in these lower temperature portions of MODE 3.

3/4.4.5 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice Inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion.

STEAM GENERATORS (Continued)

Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the primary coolant system and the secondary coolant system primary-to-secondary leakage = 1.0 gpm from both steam generators. Cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that primary-to-secondary leakage of 0.5 gpm per steam generator can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging will be required for all tubes with imperfections exceeding the plugging limit of 40% of the tube nominal wall thickness. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the Reactor Coolant Pressure Boundary. These detection systems are consistent with the recommendations of

SPECIFIC ACTIVITY (Continued)

The sample analysis for determining the gross specific activity and $\bar{E}$ can exclude the radioiodines because of the low primary coolant limit of 1 microcurie/gram DOSE EQUIVALENT I-131, and because, if the limit is exceeded, the radioiodine level is to be determined every 4 hours. If the gross specific activity level and radioiodine level in the primary coolant were at their limits, the radioiodine contribution would be approximately 1%. In a release of primary coolant with a typical mixture of radioactivity, the actual radioiodine contribution would probably be about 20%. The exclusion of radionuclides with half-lives less than 10 minutes from these determinations has been made for several reasons. The first consideration is the difficulty in identifying short-lived radionuclides in a sample that requires a significant time to collect, transport, and analyze. The second consideration is the predictable delay time between the postulated release of radioactivity from the primary coolant to its release to the environment and transport to the SITE BOUNDARY, which is relatable to at least 30 minutes decay time. The choice of 10 minutes for the half-life cutoff was made because of the nuclear characteristics of the typical primary coolant radioactivity. The radionuclides in the typical primary coolant have half-lives of less than 4 minutes or half-lives of greater than 14 minutes, which allows a distinction between the radionuclides above and below a half-life of 10 minutes. For these reasons the radionuclides that are excluded from consideration are expected to decay to very low levels before thay could be transported from the primary coolant to the SITE BOUNDARY under any accident condition.

Based upon the above considerations for excluding certain radionuclides from the sample analysis, the allowable time of 2 hours between sample taking and completing the initial analysis is based upon a typical time necessary to perform the sampling, transport the sample, and perform the analysis of about 90 minutes. The gross count should be made in a reproducible geometry of sample and counter having reproducible γ or β self-shielding properties. The counter should be reset to a reproducible efficiency versus energy. It is not necessary to identify specific nuclides.

The determination of the contributors to the $\bar{E}$ result should be based upon those energy peaks identifiable with a 95% confidence level. The radiochemical determination of nuclides should be based on multiple counting of the sample with typical counting basis following sampling of less than 1 hour, about 2 hours, about 1 day, about 1 week, and about 1 month.

Reducing T_{avg} to less than 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves. The surveillance requirements provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take corrective action. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 5.2 of the FSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

During heatup, the thermal gradients through the reactor vessel wall produce thermal stresses which are compressive at the reactor vessel inside surface and which are tensile at the reactor vessel outside surface. Since reactor vessel internal pressure always produces tensile stresses at both the inside and outside surface locations, the total applied stress is greatest at the outside surface location. However, since neutron irradiation damage is larger at the inside surface location when compared to the outside surface, the inside surface flaw may be more limiting. Consequently, for the heatup analysis both the inside and outside surface flaw locations must be analyzed for the specific pressure and thermal loadings to determine which is more limiting.

During cooldown, the thermal gradients through the reactor vessel wall produce thermal stresses which are tensile at the reactor vessel inside surface and which are compressive at the reactor vessel outside surface. Since reactor vessel internal pressure always produces tensile stresses at both the inside and outside surface locations, the total applied stress is greatest at the inside surface location. Since the neutron indication damage is also greatest at the inside surface location the inside surface flaw is the limiting location. Consequently, only the inside surface flaw must be evaluated for the cooldown analysis.

The heatup and cooldown limit curves Figures 3.4-2, 3.4-3 and 3.4-4 are composite curves which were prepared by determining the most conservative case, with either the inside or outside wall controlling, for any heatup rate of up to 50 degress F per hour or cooldown rate of up to 100 degrees F per hour. The heatup and cooldown curves were prepared based upon the most limiting value of the predicted adjusted reference temperature at the end of the service period, and they include adjustments for possible errors in the pressure and temperature sensing instruments.

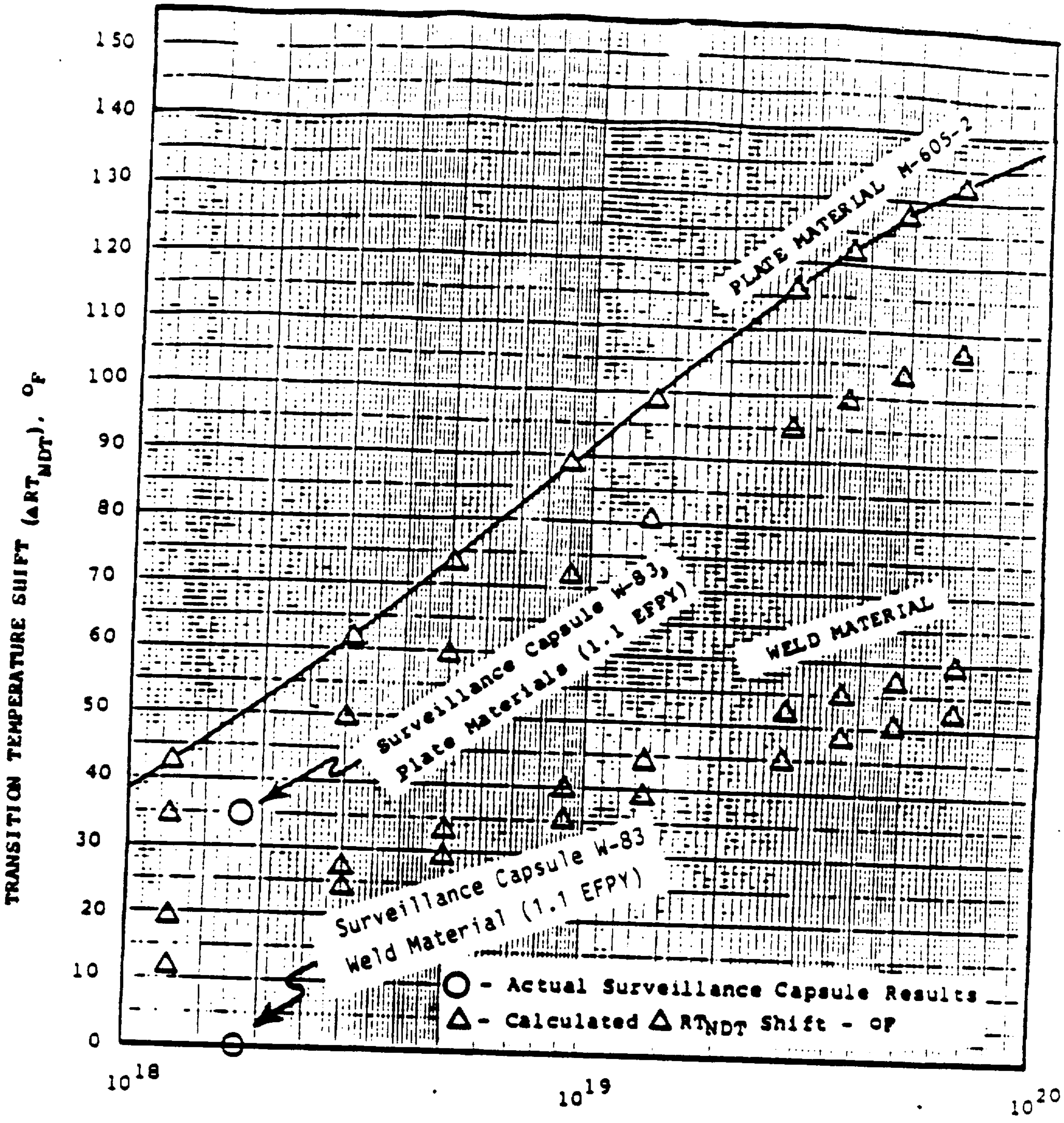
The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these tests are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E greater than 1 MeV) irradiation will cause an increase in the RT_{NDT}. An adjusted reference temperature can be predicted using a) the initial RT_{NDT}, b) the fluence (E greater than 1 MeV), including appropriate adjustments for neutron attenuation and neutron energy spectrum variations through the wall thickness, c) the copper and nickel contents of the material, and d) the transition temperature shift from the curve shown in Figure B 3/4.4-1 as recommended by Regulatory Guide 1.99, Revision 2 (Draft), "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials." The heatup and cooldown limit curves Figures 3.4-2, 3.4-3 and 3.4-4 include predicted adjustments for this shift in RT_{NDT} at the end of the applicable service period.

TABLE B 3/4.4-1

REACTOR VESSEL TOUGHNESS

Piece No.	Code No.	Material	Vessel Location	Drop Weight Results	Temperature of Charpy V-Notch RT (°F) NDT @ 50 ft - lb	Minimum Upper Shelf Cv energy for Longitudinal Direction Charpy[1] Ft - lb
122-102A	M-604-1	SA 533B Cl 1	Upper Shell Plate	0	+50	-
122-102B	M-604-2	SA 533B Cl 1	Upper Shell Plate	+10	+50	-
122-102C	M-604-3	SA 533B Cl 1	Upper Shell Plate	-10	+10	-
124-102B	M-605-1	SA 533B Cl 1	Intermediate Shell Plate	0	+30	105
124-102C	M-605-2	SA 533B Cl 1	Intermediate Shell Plate	-10	+10	113
124-102A	M-605-3	SA 533B Cl 1	Intermediate Shell Plate	-20	0	113
142-102C	M-4116-1	SA 533B Cl 1	Lower Shell Plate	-30	+20	91
142-102B	M-4116-2	SA 533B Cl 1	Lower Shell Plate	-50	+20	105
142-102A	M-4116-3	SA 533B Cl 1	Lower Shell Plate	-40	+20	100
102-101	M-4110-1	SA 533B Cl 1	Closure Head	-10	+30	-
106-101	M-4101-1	SA 508 Cl 2	Closure Head Flange	0	0	-
128-101A	M-4102-1	SA 508 Cl 2	Inlet Nozzle	-20	-20	-
128-101D	M-4102-2	SA 508 Cl 2	Inlet Nozzle	-20	-20	-
128-101B	M-4102-3	SA 508 Cl 2	Inlet Nozzle	0	0	-
128-101C	M-4102-4	SA 508 Cl 2	Inlet Nozzle	-10	-10	-
128-301B	M-4103-1	SA 508 Cl 2	Outlet Nozzle	-20	-20	-
128-301A	M-4103-2	SA 508 Cl 2	Outlet Nozzle	-30	-30	-
126-101	M-602-1	SA 508 Cl 2	Vessel Flange	-30	-10	-
131-102A	M-4104-1	SA 508 Cl 1	Inlet Nozzle Safe End	-20	+20	-
131-102D	M-4104-2	SA 508 Cl 1	Inlet Nozzle Safe End	-20	+20	-
131-102B	M-4104-3	SA 508 Cl 1	Inlet Nozzle Safe End	-20	+20	-
131-102C	M-4104-4	SA 508 Cl 1	Inlet Nozzle Safe End	-20	+20	-
131-101B	M-4105-1	SA 508 Cl 1	Outlet Nozzle Safe End	-10	0	-
131-101A	M-4105-2	SA 508 Cl 1	Outlet Nozzle Safe End	-10	0	-
152-101	M-4112-1	SA 533B Cl 1	Bottom Head Dome	-50	-40	-
154-102 (A to F)	M-4111-1	SA 533B Cl 1	Bottom Head Torus	-40	+40	-
104-102 (A to D)	M-4109-1	SA 533B Cl 1	Closure Head Torus	-60	-10	-

(1) Reported only for beltline region plates.



ST. LUCIE UNIT 2

Figure B 3/4.4-1

Reference Temperature (RT_{NDT}) increases as a function of fast (E > 1 MeV) neutron fluence (550°F irradiation) for Reactor Vessel Beltline Materials

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185-73 and 10 CFR Appendix H, reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. The surveillance specimen withdrawal schedule is shown in Table 4.4-5. Since the neutron spectra at the irradiation samples and vessel inside radius are essentially identical, the measured transition shift for a sample can be applied with confidence to the adjacent section of the reactor vessel. The heatup and cooldown curves must be recalculated when the delta RT_{NDT} determined from the surveillance capsule is different from the calculated delta RT_{NDT} for the equivalent capsule radiation exposure. The lead factors shown in Table 4.4-5 are the ratio of neutron flux at the surveillance capsule to that at the reactor inside surface.

The pressure-temperature limit lines shown on Figures 3.4-2, 3.4-3 and 3.4-4 for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR 50.

The maximum RT_{NDT} for all Reactor Coolant System pressure-retaining materials, with the exception of the reactor pressure vessel, has been determined to be 60°F. The Lowest Service Temperature limit line shown on Figures 3.4-2, 3.4-3 and 3.4-4 is based upon this RT_{NDT} since Article NB-2332 (Summer Addenda of 1972) of Section III of the ASME Boiler and Pressure Vessel Code requires the Lowest Service Temperature to be RT_{NDT} + 100°F for piping, pumps, and valves. Below this temperature, the system pressure must be limited to a maximum of 20% of the system's hydrostatic test pressure of 3125 psia.

The limitations imposed on the pressurizer heatup and cooldown rates and spray water temperature differential are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code requirements.

The OPERABILITY of two PORVs, two SDCRVs or an RCS vent opening of greater than 3.58 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold leg temperatures are less than or equal to the applicable maximum LTOP temperatures. The Low Temperature Overpressure Protection System has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either (1) a safety injection actuation in a water-solid RCS with the pressurizer heaters energized or (2) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 40°F above the RCS cold leg temperatures with the pressurizer water-solid.

3/4.4.10 REACTOR COOLANT SYSTEM VENTS

Reactor Coolant System vents are provided to exhaust noncondensible gases and/or steam from the primary system that could inhibit natural circulation core cooling. The OPERABILITY of at least one Reactor Coolant System vent path from the reactor vessel head and the pressurizer steam space ensures the capability exists to perform this function.

The redundancy design of the Reactor Coolant System vent systems serves to minimize the probability of inadvertent or irreversible actuation while ensuring that a single failure of a vent valve, power supply, or control system does not prevent isolation of the vent path.

The function, capabilities, and testing requirements of the Reactor Coolant System vent system are consistent with the requirements of Item II.b.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

3/4.4.11 STRUCTURAL INTEGRITY

The inservice inspection and testing programs for ASME Code Class 1, 2, and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. These programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR Part 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR Part 50.55a (g) (6) (i).

Components of the reactor coolant system were designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code 1971 Edition and Addenda through Summer 1973.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 31

TO FACILITY OPERATING LICENSE NO. NPF-16

FLORIDA POWER & LIGHT COMPANY, ET AL.

ST. LUCIE PLANT, UNIT NO. 2

DOCKET NO. 50-389

BACKGROUND

By letter dated November 27, 1987, Florida Power and Light Company (FPL, the licensee) proposed changes to the Technical Specifications (TS) for the St. Lucie Plant, Unit 2. The proposed changes would revise Pressure/Temperature (P/T) limits, define a new Reactor Coolant System (RCS) cold leg temperature range for Low Temperature Overpressure Protection (LTOP) and establish the minimum cold leg temperature for Power Operated Relief Valve (PORV) use for LTOP. Additional information was submitted by letters dated May 4 and 20, 1988.

EVALUATION

The licensee has generated new P/T limit curves for six operating periods starting at 4 Effective Full Power Years (EFPY) out to 32 EFPY. The licensee has also performed a new LTOP analysis to ensure that the reactor coolant pressure boundary (RCPB) integrity will be maintained in the low temperature mode of operation in accordance with the new P/T limits during each of the operating periods.

The proposed changes to the St. Lucie Unit 2 Technical Specifications would modify Sections 1.16, 3.4.1.3, 3.4.1.4.1, 3.4.4, 3.4.9.1, 4.4.9.1.2, 3.4.9.3, 4.4.9.3.1, Bases 3/4.4.1, 3/4.4.4, 3/4.4.9, Figures 3.4-2, 3.4-3, B 3/4.4-1, and Tables 4.4-5, and B 3/4.4-1.

A. Reactor Coolant System (RCS) P/T Limits

Appendix G, 10 CFR Part 50 specifies fracture toughness requirements for ferritic materials of pressure-retaining components of the RCPB of light water nuclear power reactors to provide adequate margins of safety during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests, to which the pressure boundary may be subjected over its service lifetime. This Appendix states that the requirements of the ASME Code, as supplemented by Appendix G, 10 CFR Part 50, must be met. Appendix G to Section III, Division 1, of the ASME Code presents a procedure, based on linear elastic fracture mechanics, for obtaining the allowable loadings for ferritic pressure retaining materials in components. The procedure postulates flaws on the surfaces of components and relates the allowable loadings to the reference nil ductility temperature (RT_{NDT}), defined in the ASME Code.

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The RT_{NDT}, adjusted to account for the effects of neutron radiation embrittlement during service, may be calculated either by the method outlined by the staff in Regulatory Guide 1.99, Revision 2 (Draft), "Radiation Embrittlement of Reactor Vessel Materials" or by a method proposed by the licensee, provided that the licensee demonstrates to the staff that the proposed method provides at least as good a correlation of data on neutron embrittlement of the low-alloy steels currently used for light water nuclear reactor vessels as does the draft Regulatory Guide.

Standard Review Plan Section 5.3.2, NUREG-0800, Revision 1, July 1981 is used by the staff to evaluate the acceptability of the P/T limits for the reactor vessel.

The reactor vessel inside radius is 86.220 inches and its wall thickness is 8.625 inches. Due to its copper and nickel content (.13 and .62 weight percent, respectively), its initial RT_{NDT} (10 degrees F) and its location relative to the reactor core (the beltline region of the vessel), the base metal of intermediate shell plate, M605-2, will at any given time throughout the rest of vessel life exhibit the largest adjusted RT_{NDT} of any of the vessel's material and, hence, is the controlling material.

Based on the information supplied by the licensee in the application dated November 27, 1987, the staff does not find the licensee's P/T limits curves acceptable as presented. The reason for this, as discussed below, is that the staff does not accept the licensee's analytical method. Consequently, this report focuses on P/T limits curves for the first of the proposed six operating periods, i.e., those for the interval 4 to 10 EFPY.

In computing the P/T limits curves the licensee has calculated the maximum fluence (E>1MeV) and the maximum adjusted RT_{NDT} to be 0.97 x 10^{19} n/cm^2 and 135 degrees F, respectively, at 1/4t from the vessel wall inner surface (where t refers to vessel thickness) and 0.22 x 10^{19} n/cm^2 and 98 degrees F, respectively, at 3/4t from the wall inner surface, based on a wall inner surface fluence of 1.8 x 10^{19} n/cm^2. The staff has calculated the maximum "adjusted" fluence (E>1MeV) and the maximum adjusted RT_{NDT} at the same locations within the wall to be 1.07 x 10^{19} n/cm^2 and 137 degrees F, and 0.38 x 10^{19} n/cm^2 and 111 degrees F, respectively, based on the same wall inner surface fluence. The reason for the difference is that the licensee chose to deviate from the method described in Regulatory Guide 1.99, Revision 2 (Draft).

The licensee used a Discrete Ordinates Transport model of the reactor core to vessel wall configuration to calculate peak fluences (E>1MeV) at the vessel inner surface and through the vessel wall. The core power distribution was chosen to bound future fuel management strategies with respect to the fluences accumulated in the vessel wall. Fluences thus calculated were substituted into equation (2) from Regulatory Guide 1.99, Revision 2 (Draft) to obtain the irradiation induced shifts in RT_{NDT} and equation (1) was used to yield the adjusted RT_{NDT} values.

The staff used equation (3) together with equations (2) and (1) of Regulatory Guide 1.99, Revision 2 (Draft) to calculate the adjusted RT_{NDT} values. Equation (3) gives the "effective" or "adjusted" fluence (E>1MeV) as a function of distance from the inner surface into the wall, given the fluence (E>1MeV) at the wall inner surface. Equation (3) is based on a "typical" neutron spectrum at the inner surface and takes into account the effect of spectral hardening with distance into the wall. The significance of this is that even in the energy range E>1MeV, higher energy neutrons produce more atomic displacements in the wall and, hence, make a larger contribution to embrittlement and RT_{NDT} shift than do lower energy neutrons. Equation (3) is a way of approximating a more complex spatial and energy dependent parameter, neutron spectrum (which would have to be used with energy dependent displacement cross sections), in terms of a simple "adjusted" spatial variation of the historical correlation parameter for neutron irradiation effects data, fluence (E>1MeV).

It is generally recognized that the best parameter for correlating data on changes in the properties of metals and alloys due to neutron irradiation is displacements per atom (dpa), calculated using the actual neutron spectra the material is exposed to during irradiation and the displacement cross section for the material, as discussed in ASTM Standard E693-79. The staff acknowledges this and describes in Regulatory Guide 1.99, Revision 2 (Draft) the way to modify Equation (3) using calculations of dpa to yield the best calculated values of RT_{NDT}.

As a result of the licensee's use of fluence, instead of an adjusted fluence, the licensee's values of adjusted RT_{NDT} at 1/4t and, particularly, 3/4t are not conservative compared to those obtained using the draft Regulatory Guide.

In the following discussion of the P/T limits curves, there is no discussion of the effects of LTOP controlling pressures; these will be discussed specifically in the next section of this Safety Evaluation.

For inservice hydrostatic testing, the postulated flaw at 1/4t is controlling and the P/T limits curve calculated by the staff is essentially the same as that calculated by the licensee, as shown in Figure 3.4-3, and the resulting minimum criticality temperature of 270 degrees F is acceptable to the staff. Similarly, for cooldown, the postulated flaw at 1/4t is controlling. The cooldown curve for 100 degrees F per hour computed by the licensee conservatively bounds the corresponding P/T limits curve computed by the staff. Also, the cooldown curves for 0 (isothermal), 30, and 50 degrees F per hour, all shown in Figure 3.4-3, are conservative and acceptable to the staff.

The licensee proposes to add Figure 3.4-4 showing the maximum allowable cooldown rate versus reactor coolant temperature. Following this curve keeps the reactor coolant pressure above 345 psia until the minimum boltup temperature is reached. The 345 psia is an LTOP controlling pressure which will be discussed further in the next section of this report. This addition is acceptable to the staff.

For heatup, the postulated flaw at 3/4t is controlling for a heatup rate of 50 degrees F per hour. At this heatup rate the curve computed by the licensee does not bound the corresponding P/T limits curve calculated by the staff. As to be expected, the minimum temperature versus pressure curve calculated by the licensee for this heatup rate is 13 degrees F lower than that calculated by the staff. The difference is due to the above discussed difference in the adjusted RT_{NDT} calculated by the licensee and that calculated by the staff, i.e., 13 degrees F at 3/4t. The P/T limit curve for a heatup rate of 50 degrees F per hour as shown in Figure 3.4-2 (for which the RT_{NDT} at 3/4t was taken to be 98 degrees F) corresponds more closely to the 50 degrees F per hour heatup curve that results from using the entire methodology of Regulatory Guide 1.99, Revision 2 (Draft), with a wall inner surface fluence (E>1MeV) of 1.1×10^{19} n/cm^2, which is equivalent to 6 EFPY. Thus, the curves in Figures 3.4-2, 3.4-3 and 3.4-4, which were drawn for 10 EFPY, are acceptable to the staff to only 6 EFPY. The P/T limits curve for heatup and cooldown while the reactor is critical is obtained by adding 40 degrees F to the limiting heatup curve in Figure 3.4-2 or cooldown curve in Figure 3.4-3, where the minimum critical temperature is the minimum permissible temperature for the inservice system hydrostatic test pressure, 270 degrees F, as discussed above. The staff finds the core critical P/T limits curve, as shown in Figure 3.4-2, acceptable out to 6 EFPY. Correspondingly, the licensee's changes to LCO 3.4.9.1 and Surveillance Requirement 4.4.9.1.2 are acceptable only for Figures 3.4-2, 3.4-3 and 3.4-4, out to 6 EFPY.

This was discussed with the licensee. By letter dated May 20, 1988, the licensee requested that P/T limit curves valid to 6 EFPY be issued in the interim. The staff agreed to conduct a detailed review, separate from this amendment, of the Combustion Engineering Report submitted by FPL with the November 27, 1987 application to determine whether it justifies the FPL method for predicting delta RT_{NDT}.

The lowest service temperature, 168 degrees F, the maximum boltup temperature, 80 degrees F, and the maximum service pressure, 550 psia, are considered conservative and acceptable to the staff.

The licensee proposes to change Figure B 3/4.4-1, Nil-Ductility Transition Temperature Increase As A Function of Fast (E>1MeV) Neutron Fluence (550 degrees F, Irradiation), to show a more realistic projection of transition temperature shift than that currently in the TS. A staff modification to Bases 3/4.4.9 which appropriately discusses the new figure, in particular the fluence, in terms of the proper interpretation and use of Regulatory Guide 1.99, Revision 2 (Draft) was discussed with the licensee. The proposed change, taken together with the modified Bases, are acceptable to the staff. Other changes to this Bases proposed by the licensee are clarifications and are acceptable to the staff.

The licensee also proposes to change Table 4.4-5, Reactor Vessel Material Surveillance Program - Withdrawal Schedule, increasing the lead factor from 1.15 to <=1.5, and Table B 3/4.4-1, Reactor Vessel Toughness, correcting errors in the columns Drop Weight Results and Temperature of Charpy V-Notch RT_{NDT} (degrees F) @ 50 ft-lb. These changes are acceptable to the staff.

B. Modifications to LTOP

The LTOP system provides assurance that 10 CFR Part 50, Appendix G limits will not be violated during both normal operation and overpressurization events due to equipment malfunction or operator error provided the administrative controls specified by the licensee are implemented. The current LTOP system hardware and setpoints were assumed to be effective up to the end of license, which is currently at the end of 32 EFPY. In the design of the LTOP system, mass addition and energy addition overpressurization events were considered. The associated design basis pressure transients were analyzed based on the currently utilized means for transient mitigation that include the shutdown cooling relief valves (SDCRVs) and the power operated relief valves (PORVs). The pressure transient analysis results were evaluated with the new P/T limits, discussed above, to yield a number of administrative and operational limitations to be implemented to ensure adequate LTOP. These changes affect relief valve alignment temperatures, heatup and cooldown rates and requirements for reactor coolant pump (RCP) operation.

Standard Review Plan Section 5.2.2, NUREG-0800, Revision 2, July 1981 and Branch Technical Position RSB 5-2, NUREG-0800, Revision 0, July 1981 are used by the staff to evaluate the acceptability of LTOP.

The LTOP system makes use of two SDCRVs at lower RCS temperatures, two PORVs through the remaining part of the LTOP temperature range and an RCS vent. According to the licensee's analysis, two mass addition events (actuation of a single HPSI pump with all three charging pumps operating, and actuation of two HPSI pumps with all three charging pumps operating) and an energy addition event (reactor coolant pump start, with a positive secondary-to-primary temperature differential) were the controlling LTOP system design basis events. The licensee has concluded that no modifications to the LTOP system hardware or setpoints are needed for operation beyond 4 EFPY; the current lift setting for the PORVs is 470 psia, for the SDCRVs is 350 psia and the vent area is 3.58 in^2.

As a result of the energy addition transient analysis, assuming a secondary-to-primary temperature differential of 40 degrees F, the maximum pressurizer pressures were determined to be 535 psia for RCP start with PORV actuation and 343 psia for RCP start with SDCRV actuation. As a result of the mass addition transient analysis, the maximum pressures for two HPSI and three charging pumps were determined to be 535 psia for PORV actuation and 355 psia for SDCRV actuation, while the maximum pressures for one HPSI and three charging pumps were determined to be 492 psia for PORV actuation and 345 psia for SDCRV actuation.

The results of the above transient analyses have been previously accepted by the staff and since the hardware and setpoints remain unchanged, they are applicable beyond 4 EFPY. However, since the P/T limits change due to neutron embrittlement of the vessel wall, the LTOP valve alignment temperatures and heatup and cooldown rates are affected.

An LTOP controlling pressure identifies an RCS pressure limit which should not be exceeded during any overpressurization event that could occur in the corresponding temperature region while mitigated by the LTOP system. When applied to the P/T limits curves, an LTOP controlling pressure also provides a lower bound pressure limit for these curves; i.e., a controlling pressure is more limiting than the P/T limits curves above it. In the PORV-mitigated transients, the controlling pressure was assumed to equal the highest maximum transient pressure of 535 psia that is applicable at all RCS temperatures at which LTOP is provided by the PORVs. In the SDCRV-mitigated transients, the controlling pressure was taken to be the "second highest maximum pressure," 345 psia, since Technical Specifications require one HPSI to be rendered inoperable at coolant temperatures less than or equal to 200 degrees F and no intersections exist between a horizontal line corresponding to the maximum pressure associated with SDCRV-mitigated transients, 355 psia, and any P/T limits curve for temperatures greater than 200 degrees F. The staff has previously found these control pressures acceptable.

The intersection between the control pressure 535 psia and the 50 degrees per hour heatup curve in Figure 3.4-2 gives the minimum temperature at which the LTOP function can be transferred from the SDCRVs to the PORVs during heatup, 156 degrees F, and the intersection between 535 psia and the 100 degrees per hour. cooldown curve in Figure 3.4-3 gives the minimum temperature at which the LTOP function must be transferred from the PORVs to the SDCRVs during cooldown, 179 degrees F. The temperatures at which cooldown rates must be reduced are determined by the control pressure 345 psia. These cooldown rates versus coolant temperature are plotted in Figure 3.4-4. The intersections between the safety valve setpoint of 2500 psia and the 50 degree per hour heatup curve and the 100 degree per hour cooldown curve define the maximum LTOP temperatures for heatup, 313 degrees F, and for cooldown, 304 degrees F; these temperatures are taken as upper bounds for the regions in which LTOP is required. The staff finds these limiting temperatures, shown in Tables 3.4-3 and 3.4-4, acceptable to only 6 EFPY for the reasons discussed in the previous section of this report.

The licensee has proposed changes to Limiting Condition for Operation (LCO) 3.4.9.3 and associated Action Statements and Surveillance Requirements. The LCO would state that unless the RCS is depressurized and vented by at least 3.85 in^2, at least one of the following overpressure protection devices shall be operable: a) two PORVs with their associated block valves open; b) two SDCRVs; or c) one PORV with its associated block valve open in conjunction with one SDCRV (with the above-mentioned PORV and SDCRV lift settings). This would make the LCO somewhat more flexible, but still as conservative as before, and is acceptable to the staff. The change to Surveillance Requirement 4.4.9.3.1 is a clarification and is also acceptable to the staff. However, the staff does not accept the proposed change to Action Statement b) since the licensee did not provide adequate justification for the additional 32-hour delay in the proposed action. The staff has reworded this section to read: "b. With none of the overpressure protection devices being used for LTOP OPERABLE, within the next eight hours either: 1. Restore at least one overpressure protection device to OPERABLE status or vent the RCS; or 2. Be at a temperature above the LOW TEMPERATURE RCS OVERPRESSURE PROTECTION RANGE of Table 3.4-3."

The licensee's proposed changes to LCO 3.4.1.3, LCO 3.4.1.4.1, and LCO 3.4.4 make one clarification and reference Table 3.4-3, Low Temperature RCS Overpressure Protection Range. These changes are acceptable to the staff.

The licensee's proposed changes to Bases 3/4.4.1 and 3/4.4.4 are to specify the low temperature RCS overpressure protection range and the minimum cold leg temperature for PORV use for LTOP during heatup and cooldown and provide clarifications of interpretation. The staff finds these changes, along with the proposed change to Definition 1.16, acceptable.

FINDINGS

The modifications to the Technical Specifications proposed in this amendment by FPL, for the St. Lucie Plant, Unit 2, concerning the RCS P/T limits and LTOP, are judged by the NRC staff to be adequate and acceptable, with the exceptions discussed above, out to 6 EFPY. This will permit plant operation for an additional 2 EFPY, while the staff and licensee resolve the issues discussed above.

EXIGENT CIRCUMSTANCES

The Commission's regulations, 10 CFR 50.91, contain provisions for issuance of amendments when the usual 30-day public notice period cannot be met. One type of special exception is an exigency. An exigency is a case where the staff and licensee need to act promptly, but failure to act promptly does not involve a plant shutdown, derating, or delay in startup. The exigency case usually represents an amendment involving a safety enhancement to the plant.

Under such circumstances, the Commission notifies the public in one of two ways: by issuing a Federal Register notice providing an opportunity for hearing and allowing at least two weeks for prior public comments, or by issuing a press release discussing the proposed changes, using the local media. In this case, the Commission used the first approach.

The licensee submitted the request for amendment on November 27, 1987. It was noticed in the Federal Register on February 10, 1988 (53 FR 3954), at which time the staff proposed a no significant hazards consideration determination. The licensee requested that the amendment be issued prior to May 13, 1988, at which time the plant was expected to reach 4 EFPY. The licensee subsequently advised the staff by letter dated May 4, 1988 that the plant would reach 4 EFPY no earlier than May 23, 1988. The "no earlier than" phrase was used because if the plant tripped off line or was brought down temporarily for some other reason, the 4.0 EFPY would occur at a date later than May 23.

The staff's review of certain portions of the licensee's submittal is not complete at this time. By letters dated May 4 and 20, 1988, the licensee requested the staff to issue P/T limits valid until 6 EFPY, in order to permit the staff to complete its review and not affect plant operation. In essence, the originally proposed curves which would have been valid until 10 EFPY will be changed to be valid until 6 EFPY. The staff renoticed the amended application on May 27, 1988 (53 FR 19357) because it represented a significant change from what was previously noticed. The staff proposed to determine that the amended application involves no significant hazards considerations. The net effect of the change is a more restrictive set of Technical Specifications.

Therefore, the staff is issuing the amendment under exigent circumstances. The existing P/T limits expired on May 23, 1988. The staff and the licensee have been evaluating the issues associated with the review and currently agree that the 6 EFPY curves are satisfactory. The licensee did not request emergency treatment of the amended application; the staff does not believe that an emergency situation exists. However, the staff does believe that the amendment should be issued promptly.

There were no public comments in response to the either notices published in the Federal Register.

FINAL NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The Commission's regulations in 10 CFR 50.92 state that the Commission may make a final determination that a license amendment involves no significant hazards considerations if operation of the facility in accordance with the amendment would not: (1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

Operation of the facility in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated. Appendix G to 10 CFR Part 50 specifies fracture toughness requirements for ferritic materials of pressure-retaining components of the reactor coolant pressure boundary of the light-water nuclear power reactors to provide adequate margins of safety during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests, which the pressure boundary may be subjected to over its service lifetime. One acceptable method used by the staff to assure that a license meets Appendix G is contained in Regulatory Guide 1.99, Revision 2 (Draft), entitled "Radiation Embrittlement of Reactor Vessel Materials." The staff used the Regulatory Guide and determined that the proposed 10 EFPY pressure-temperature limit curves are adequate and acceptable out to 6 EFPY, and that the associated LTOP settings should be based on the 6 EFPY P/T limit curves. This determination was based on the staff's analysis, as described in the Evaluation section of this Safety Evaluation; it consisted, in part, of independent calculations. Thus, the staff concludes that the curves are acceptable up to 6 EFPY, the licensee will meet Appendix G of 10 CFR Part 50, and that the use of the curves for RCS heating and cooldown and LTOP mitigation will not involve a significant increase in the probability or consequences of an accident previously evaluated.

Operation of the facility in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated. All design basis energy addition and mass addition transients have been evaluated and there is no significant change in the configuration or operation of the facility due to the proposed amendment. Thus, there is no new or different kind of accident created as a result of this amendment.

Operation of the facility in accordance with the amendment will not involve a significant reduction in a margin of safety. Appendix G to 10 CFR Part 50 recognizes that the P/T limits and associated LTOP instrument settings must be re-evaluted periodically to ensure safety margin because of foreseen radiation damage that occurs to the reactor pressure vessel over its lifetime. The new limit curves maintain the margin of safety and meet Appendix G to 10 CFR Part 50. Thus, there is no reduction in the margin of safety.

Based upon the above considerations, the staff concludes that the amendment meets the three criteria of 10 CFR 50.92. Therefore, the staff has made a final determination that the proposed amendment does not involve a significant hazards consideration.

ENVIRONMENTAL CONSIDERATIONS

This amendment involves changes in the installation or use of the facilities components located within the restricted area as defined in 10 CFR 20 or changes to a surveillance requirement. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of an accident of a type different from any evaluated previously, and does not involve a significant reduction in a margin of safety, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: June 14, 1988

Principal Contributor:

John O. Schiffgens